UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

December 18, 2009

Mr. Joseph E. Pollock
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 3 – NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT MODIFICATIONS TEAM INSPECTION REPORT 05000286/2009006

Dear Mr. Pollock:

On November 5, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 3. The enclosed inspection report documents the inspection results, which were discussed on November 5, 2009, with Mr. P. Conroy and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. In conducting the inspection, the team reviewed selected procedures, calculations and records, observed activities, and interviewed station personnel.

Based on the results of this inspection, no findings of significance were identified.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Lawrence T. Doerflein

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-286
License No. DPR-64

Enclosure: Inspection Report No. 05000286/2009006
w/ Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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/RA/
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OFFICE	RI/DRS		RI/DRP		RI/DRS					
NAME	JSchoppy/jgs*		MGray/mkg*		LDoerflein LTD					
DATE	11/16/09		12/07/09		12/18/09					

*see prior concurrence

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-286
License No.:	DPR-64
Report No.:	05000286/2009006
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating Unit 3
Location:	450 Broadway, GSB Buchanan, NY 10511-0249
Inspection Period:	October 19 – November 5, 2009
Inspectors:	J. Schoppy, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader M. Halter, Reactor Inspector, DRS P. McKenna, Reactor Inspector, DRS
Approved By:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000286/2009006; 10/19/2009 – 11/5/2009; Indian Point Generating (Indian Point) Unit 3; Engineering Specialist Plant Modifications Inspection.

This report covers a two week on-site inspection period of the evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three region based engineering inspectors. No findings of significance were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (28 samples)

a. Inspection Scope

The team reviewed three safety evaluations to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance with 10 CFR 50.59 requirements. In addition, the team evaluated whether Entergy had been required to obtain NRC approval prior to implementing the change. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications (TSs), and plant drawings, to assess the adequacy of the safety evaluations. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Evaluations," as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The team also reviewed a sample of twenty five 10 CFR 50.59 screenings and applicability determinations for which Entergy had concluded that no safety evaluation was required. These reviews were performed to assess whether Entergy's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample included design changes, calculations, procedure changes, and setpoint changes.

The team reviewed the safety evaluations that Entergy had performed during the time period covered by this inspection (i.e., since the last modifications inspection). The screenings and applicability determinations were selected based on the safety significance, risk significance and complexity of the change to the facility.

In addition, the team compared Entergy's administrative procedures used to control the screening, preparation, review, and approval of safety evaluations to the guidance in NEI 96-07 to determine whether those procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations, screenings, and applicability determinations are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications (9 samples)

.2.1 Residual Heat Removal Pump Motor Flood Protection

a. Inspection Scope

The team reviewed modification 5000041988 that was designed to protect Unit 3 residual heat removal (RHR) pump motors from damage due to flooding of the primary auxiliary building (PAB). The modification installed a float operated valve in the PAB with an attached drain pipe that penetrated the exterior wall of the PAB and was routed below grade to a nearby storm drain manhole in the transformer yard.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the RHR system and the adjacent structures, systems, and components (SSCs) had not been degraded by the PAB penetration and installed float valve. The team reviewed the calculations for the sizing and placement of the float operated valve and for the associated piping to verify the adequacy of the design. The team also reviewed whether Entergy updated the applicable drawings to incorporate the modification. The team reviewed the post modification test (PMT) results to verify that the valve would function in accordance with the design assumptions. The team also conducted a walk down of the modification to determine if the valve would function in accordance with technical and design assumptions and to verify that Entergy maintained adequate configuration control. Additionally, the team reviewed the 10 CFR 50.59 screen and engineering evaluation associated with this modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.2 Surface Preparation Work on Pressurizer Surge Nozzle Dissimilar Welds Prior to Examination

a. Inspection Scope

The team reviewed a modification (Engineering Change (EC) No. 9220) which performed surface preparation on the Unit 3 pressurizer surge nozzle dissimilar welds prior to in-service inspection (ISI). The modification also removed four metal lugs from the nozzle safe end. The surface preparation and lug removal was necessary to attain sufficient coverage of the examination volume for analyzing the dissimilar metal weld condition through non-destructive examination (NDE) methods.

The team assessed whether the modification was consistent with assumptions in the design and licensing bases. Additionally, the team reviewed the 10 CFR 50.59 screen associated with this modification as described in Section 1R17.1 of this report. The team reviewed engineering analyses performed to determine minimum acceptable wall thicknesses for the pressurizer surge nozzle safe end. The team interviewed ISI engineers and reviewed drawings, thickness measurements, and correspondence between Entergy and Westinghouse to verify that the as-left surge nozzle safe end condition was in accordance with design and licensing assumptions. The team verified that post-work, as-left wall thicknesses were greater than minimum acceptable to ensure the pressurizer surge line was not adversely impacted by the modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.3 Auxiliary Boiler Feedwater Pump Room Temperature Switch Setpoint Modification

a. Inspection Scope

The team reviewed the engineering change (EC No. 9070) that modified the auxiliary boiler feedwater (ABFW) pump room temperature switch alarm and trip setpoints. The temperature trip switches are designed to automatically isolate the steam supply to the turbine-driven ABFW pump during a postulated high energy line break (HELB) in the ABFW pump room. Entergy implemented this change to address a previous NRC concern associated with the ABFW pump room temperature profile and the potential for an undesired isolation of the steam supply to the turbine-driven pump during certain scenarios. The identified concern was that the steam to the turbine-driven pump would be isolated during non-HELB scenarios that credit the turbine-driven pump (see NRC Component Design Bases Inspection Report 05000247/2007007).

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the ABFW system had not been degraded by the setpoint changes. The team reviewed several related calculations associated with ABFW pump room temperature response and instrument setpoint tolerances to ensure that Entergy used conservative assumptions and appropriate inputs to adequately evaluate the modification. The team conducted several walk downs of the ABFW pump room to independently assess Entergy's configuration control, the material condition of the temperature sensors and surrounding SSCs, and the validity of Entergy's design process inputs. The team reviewed the associated PMT, several environmental qualification (EQ) files for components in the ABFW pump room, and the instrument setpoint functional test results for each of the respective temperature sensors. The team also reviewed corrective action condition reports (CRs) to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the team reviewed the 10 CFR 50.59 screen and engineering evaluation associated with this modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.4 Service Water Header Isolation Valve SWN-FCV-1112 Replacement

a. Inspection Scope

The team reviewed modification 97-3-230SWS that replaced service water (SW) header isolation valve SWN-FCV-1112 and the downstream mating flange. Entergy implemented the modification because the existing valve and mating flanges were substantially corroded. The material of the replacement valve had a higher corrosion resistance than the existing valve. Replacement valve dimensions had also changed slightly from the existing valve.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the SW system had not been degraded by the SW valve replacement. The team interviewed engineering staff, reviewed the associated PMT, and conducted a walk down of the installed valve to determine if the material condition and performance of the SW system was acceptable and in accordance with design assumptions. Additionally, the team reviewed the 10 CFR 50.59 screen and engineering evaluation associated with this modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.5 Motor Driven Auxiliary Boiler Feedwater Pump Recirculation Valve Replacements

a. Inspection Scope

The team reviewed a modification (EC No. 5155) that replaced the 31 ABFW pump's recirculation valve, BFD-53, and the 33 ABFW pump's recirculation valve, BFD-55. Entergy implemented the modification to address a history of ABFW pump recirculation valve failures. Previous failures were caused by erosion conditions resulting from cavitations within the bodies of the throttled recirculation valves.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the ABFW system had not been degraded by the modification. Additionally, the team reviewed the 10 CFR 50.59 screen associated with this modification as described in Section 1R17.1 of this report. The team reviewed calculations and technical evaluations to verify that the valves would function in accordance with design assumptions. The team reviewed the associated PMT to verify that test results appropriately supported system operability. The team conducted walkdowns of the valves to verify that the modified valves did not impact the operation of other equipment located in the vicinity of the valves and to ensure adequate configuration control. The team reviewed loading calculations and seismic evaluations to verify that installed piping and supports were adequate for the increased weight of the new valves. Finally, the team conducted interviews with engineering staff to determine if the valves would function in accordance with technical and design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.6 125V DC Breakers Replacement

a. Inspection Scope

The team reviewed modification ER-06-3-003 that replaced ten 125V DC breakers in DC distribution panel 32A with new, upgraded components. Entergy implemented the modification because the previously installed breakers were obsolete and had exceeded their recommended service life and replacement was required to prevent future component failures. The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the 125V DC system had not been degraded by the breaker replacements.

The team compared the new breaker's specifications and electrical characteristics to the old breaker's to ensure that they were electrically equivalent and that there would be no adverse impact on breaker function or coordination. The team reviewed the vendor's certification and acceptance testing results for the new breakers, Entergy's installation work order, and Entergy's PMT results. The team reviewed associated drawings and system operating procedures to ensure that Entergy had properly updated them to incorporate the changes. The team performed a visual internal inspection of distribution panel 32A to independently assess Entergy's configuration control (including breaker type and position) and the material condition of the installed breakers. The team also reviewed CRs to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the team reviewed the 10 CFR 50.59 screen and engineering evaluation associated with this modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.7 Emergency Diesel Generator Jacket Water Pressure Switch Replacements

a. Inspection Scope

The team reviewed modification 0000007135 that replaced jacket water (JW) pressure switches Nos. 3, 4 and 6 on all three emergency diesel generators (EDGs), changed the setpoints for the switches, and installed diodes across the coils of several JW circuit relays. Entergy replaced the previous EDG JW pressure switches (UE Type J54A, Model 24) with newer model switches (UE Type J54A, Model 152). Entergy made the change because the Model 152 switches have brass bellows, which makes the switch less sensitive to vibration and also improves switch reset performance characteristics. Engineering installed the diodes across the coils of the relays to provide electrical surge protection and reduce the likelihood of micro-welding of the pressure switch contacts.

The team performed this review to verify that the design bases, licensing bases and performance capability of the EDGs had not been degraded by the JW pressure switch modifications. The team verified that Entergy properly updated the associated drawings, calculations, and calibration procedures. The team interviewed engineering staff, and reviewed the associated PMT and several EDG functional tests to verify that the JW system functioned as designed. The team conducted a walk down of the accessible portions of the JW system for each of the EDGs and inspected the JW pressure switches inside the 33 EDG control panel to independently assess Entergy's design and configuration control and the material condition of this EDG support system. Additionally, the team reviewed the 10 CFR 50.59 screen and engineering evaluation associated with this modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.8 Installation of a Manway and Inside Protective Coating on 32 Emergency Diesel Generator Starting Air Tank

a. Inspection Scope

The team reviewed a modification (EC No. 6665) that installed a manway on and protective coating inside 32 EDG starting air tank. The manway provided access to the interior of the tank and facilitated the installation of the protective coating, which was applied to the interior of the tank to address historical tank internal corrosion. The corrosion previously caused restricted air flow to and impinged the turbine vanes of the EDG air start motors, causing degradation and failure.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the EDG and its air start support system had not been degraded by the modification. The team also reviewed the 10 CFR 50.59 screen associated with this modification as described in Section 1R17.1 of this report. The team interviewed engineering staff and reviewed technical evaluations associated with the modification to determine if the EDG starting air system would function in accordance with the design assumptions. The team performed several walkdowns of the starting air system, including the modified tank, and reviewed loading calculations to verify the seismic and structural qualification of the manway. The team also reviewed an engineering change notice (ECN) that changed the type of protective coating installed in the tank to verify that all potential effects on the previously performed engineering evaluations were reviewed and documented. Additionally, the team reviewed EDG surveillance data to verify that the EDG starting air system was not adversely affected by the modification. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.9 Reactor Coolant Pump Thermal Barrier Differential Pressure Transmitter Instrument Line Spool Piece Installation

a. Inspection Scope

The team reviewed modification ER-07-3-021 that installed a spool piece on a 1" instrument line connected to the high pressure side of the thermal barrier differential pressure (D/P) transmitter on the 33 reactor coolant pump (RCP). Entergy implemented this modification because the 33 RCP was returned from refurbishment with a bent 1" flanged instrument connection to PT-131. PT-131 is the 33 RCP D/P transmitter that indicates the pressure drop across the thermal barrier and labyrinth seal. This chemical and volume control system (CVCS) instrument line is rated for 2580 psig and a temperature of 650 °F and under normal operation at power contains CVCS-grade water supplied to the RCP seals from the charging pumps (~ 2300 psig and < 130 °F). This 1" instrument line is normally oriented radially outward from the RCP centerline. However, connection to the RCP flanged connection in its bent orientation resulted in interference with an adjacent cooling water line. The purpose of this modification was to design, fabricate and install a pipe spool piece that allowed connection to the PT-131 instrument line without any interference from adjacent piping.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the CVCS, RCP thermal barrier, and reactor coolant system (RCS) barrier had not been degraded by the bent line or the installed spool piece. The team reviewed the documentation supporting Westinghouse's evaluation and determination that it was acceptable to install the pump internals and use as planned (modifying the interfacing connection) with the pressure tap in the as-bent condition. The team reviewed IP-CALC-07-00071, Stress Analysis of Portion of Line 491 due to Bent RCP 33 Nozzle During 3R14, and the associated piping specifications to verify that the piping and the pipe support system containing the flanged spool piece remained within the design basis pipe stress and support limits. The team reviewed Entergy's installation work order including the associated weld specification sheets, weld maps, and completed weld data sheets. The team also reviewed Entergy's PMT results including the weld visual inspections, liquid penetrant examinations, and leakage tests. The team reviewed the associated drawings to ensure that Entergy had properly updated them to incorporate the changes. The team performed a walk down of the RCP seal parameter indications in the control room, reviewed RCP seal parameter trending data, and reviewed CRs to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the team reviewed the 10 CFR 50.59 screen associated with this modification as described in Section 1R17.1 of this report. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of CRs associated with 10 CFR 50.59 and plant modification issues to determine whether Entergy was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned or completed corrective actions were appropriate. In addition, the team reviewed CRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The CRs reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. P. Conroy, Director, Nuclear Safety Assurance, and other members of Entergy's staff at an exit meeting on November 5, 2009. The team verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel
J. Bencivenga, Design Engineering
R. Brown, Operations Procedure Writer
P. Conroy, Director, Nuclear Safety Assurance
G. Dahl, Specialist, Licensing
M. DeChristopher, System Engineering
A. De Donato, Programs and Components Engineering
B. Dolansky, Senior Lead Engineer, Code Programs Group
J. Kaczor, Design Engineering
T. McCaffrey, Manager, Design Engineering
D. Nuta, Design Engineering

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

None.

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations
08-3001-00-EVAL, Partial UFSAR Revision in Support of EC-2813 (Resolution of GSI-191), Rev.0
09-3001-00-EVAL, MDAFW Pump Branch Line Flow Imbalance, Rev. 0
09-3002-00-EVAL, TS Bases 3.6.5, Rev. 0

10 CFR 50.59 Screened-Out Evaluations
3-AOP-TURB-1, Main Turbine Trip Without a Reactor Trip, Rev. 4
3-ARP-049, Panel Local – Intake Structure, dated 3/11/07
3-COL-EL-005, Diesel Generators, Rev. 33
3-HTX-004-CCW, Component Cooling Water Heat Exchanger Maintenance, Rev. 2
3-OSP-RW-005, Service Water System Alignment for Single Service Water Header Operation, Rev. 3
3-PC-OL40, Main Steam Line Radiation Monitor Calibration, dated 12/22/08
3-PC-R73B, Revise Nuclear Instrument Intermediate Range N-36 Channel Calibration Procedure for New Tolerance Values for the Log Current Amplifier, dated 4/1/09
3-PMP-012-SWS, Service Water Pump Removal and Installation, Rev. 19
3PT-CS040, Safety Injection Pumps Suction Check Valve Operability Test, Rev. 3
3-PT-Q120B, 32 ABFP (Turbine Driven) Surveillance and IST, dated 8/31/09
3-PT-Q129, Service Water System Alignment Verification, dated 7/7/09

3-PT-R203, Visual Examination of the Reactor Vessel Head Penetrations and Head Surface for Leakage, dated 12/22/08
3-PT-Y008AR00, In-Service Pressure Test of Containment Spray (31 CSP), dated 4/30/09
3-VLV-052-RCS, Pressurizer Power Operated Relief Valve (RC-PCV-455C, RC-PCV-456) Inspection and/or Removal, Rev. 5
EC 4304, Indian Point Unit 3 Cycle 16 Reload Core Design Change, dated 3/2/09
EC5000042012, Engineering Evaluation of Vapor Containment Instrumentation, dated 4/6/07
EC 5155, Replace Throttle Valves BFD-53 & BFD-55, dated 3/6/08
EC 6665, Install Manways and Inside Protective Coating on EDG Starting Air Tanks, dated 5/12/08
EC 7622, Develop Setpoint Calculation for RWST Temperature Controller TIC-1116-S, dated 7/16/08
EC 8863, VC Sump Pump SP-313, SP-314 Require Setpoint Change and Upper Collar Design Improvements, dated 9/22/08
EC 9220, Surface Preparation Work on Unit 2 and 3 Pressurizer Surge Nozzle Dissimilar Welds Prior to Examination, dated 10/15/08
ECR 6553, Replacement of the Unit 3 Intake Trash Racks during 3R15, dated 3/2/09
ER-06-3-048/SCR-07-3-039, TM/Toxic Gas Monitoring System, dated 5/23/07
ER-07-3-021, RCP Bent Instrument Line Spool Piece, dated 3/17/07
ONOP-CVCS-3, Emergency Boration, Rev. 8

Audits and Self-Assessments
LO-IP3LO-2009-00032-CA-1, Plant Modifications and 50.59 Evaluations, dated 6/10/09
QA-04-2008-IP-1, Engineering Design Control, dated 4/25/08
QS-2008-IP-14, IPEC QA Follow-up of AFI from the Design Control Audit, dated 12/1/08

Calculations
08-00067 EDN 9508, Evaluation of Anchorage for Mounting of Existing 31, 32 & 33 EDG Starting Air Tanks Due to Addition of Manway on the Tanks, Rev. 0
08-00070, Structural Design of Pipe Stress and Support Structure for IP3 BFD 53 & 55 Valve Replacement, Rev. 0
IP3-CALC-ASC-0613, RWST Temperature Controller TIC-1116-S, Rev. 1
IP-CALC-07-00071, Stress Analysis of Portion of Line 491 due to Bent RCP 33 Nozzle During 3R14, Rev. 0
IP-CALC-07-00143, Auxiliary Feedwater Pump Room Temperature Rise (IP2), Rev. 1
IP-CALC-07-00154, Containment Atmospheric Temperature, Rev. 0
IP-CALC-07-00193, Auxiliary Feedwater Pump Room Temperature Rise (IP3), Rev. 0
IP-CALC-07-00210, Pressure and Temperature Response from High Energy Line Break in the Auxiliary Feedwater Pump Room, Rev. 0
IP-CALC-08-00031, Miscellaneous Structural Evaluation for IP2 & IP3 RHR Pump Motor Flood Protection, Rev. 0
IP-CALC-08-00061, Sizing Calculation for RHR Pump Flooding Line, Rev. 0
IP-CALC-08-00075, Emergency Diesel Generators Jacket Water Pressure Switches Setpoints, Rev. 1
IP-CALC-08-00111, Auxiliary Boiler Feedwater Pump Room Temperature Switches, Rev. 0
IP-CALC-09-00065, IP3 Steamline Break Outside Containment Auxiliary Feedwater Flow Mismatch and Resolution of IR#09-086-M004, Rev. 0

IP-CALC-09-00066, IP3 Asymmetric Auxiliary Feedwater Evaluation, Rev. 0
IP-CALC-CCW-02487, Tube Plugging Limit for CCW Heat Exchangers, Rev. 0

Completed Surveillance & Functional Tests

3-IC-N-T-1116S, Refueling Water Storage Tank Temperature Control, performed 9/8/09 & 9/10/09
3-IC-PC-I-F-625, Reactor Coolant Pump Thermal Barrier Component Cooling Header Flow, performed 3/20/09
3-PT-M079B, 32 EDG Functional Test, performed 8/13/09 & 9/8/09
3-PT-Q016, EDG and Containment Temperature SW Valves SWN-1176 & 1176A and SWN-1104 & 1105, performed 4/28/09
3-PT-R20A, Auxiliary Boiler Feed Pump Room Temperature Sensors (TC-1112A, TC-1112S), performed 5/12/09
3-PT-R20B, Auxiliary Boiler Feed Pump Room Temperature Sensors (TC-1113A, TC-1113S), performed 5/13/09

Condition Reports (CR-IP3-)

1996-0040	2008-0642	2009-1377	2009-4247	2009-4353*
2003-0165	2008-0650	2009-1401	2009-4203*	2009-4354*
2003-2127	2008-0717	2009-1436	2009-4243*	2009-4356*
2006-0145	2008-0909	2009-1560	2009-4249*	2009-4359*
2006-1317	2008-1176	2009-1760	2009-4278*	2009-4360*
2006-3511	2008-1179	2009-2155	2009-4338*	2009-4361*
2007-0164	2008-2056	2009-2329	2009-4341*	2009-4376*
2007-1226	2009-0779	2009-2343	2009-4343*	
2007-1949	2009-1016	2009-4217*	2009-4344*	
2007-2056	2009-1050	2009-4218*	2009-4345*	
2007-4552	2009-1270	2009-4219*	2009-4350	

* Condition report written as a result of inspection effort.

Design & Licensing Bases

IP3-DBD-303, Design Basis Document for the Auxiliary Feedwater System (AFWS), Rev. 3
IP3-DBD-304, Design Basis Document for the Service Water System (SWS), Rev. 3
IP3-DBD-307, Design Basis Document for 480V AC, 125V DC, 120V Vital AC Electrical Distribution System, Rev. 3
IP3-DBD-311, Design Basis Document for the Chemical and Volume Control System, Rev. 2
IP3-DBD-314, Design Basis Document for the Reactor Coolant System, Rev. 2
IP3-DBD-315, Design Basis Document for Heating, Ventilation and Air Conditioning Systems, Rev. 2
IP3-DBD-318, Design Basis Document for Seismic Building and Structures, Rev. 2
IP3-DBD-322, Design Basis Document for High Energy Line Break Outside Containment, Rev. 1
IP3-DBD-323, Design Basis Document for Containment Spray System, Rev. 1
IP3-DBD-324, Design Basis Document for the Emergency Diesel Generators and Appendix R Diesel Generator, Rev. 1
NL-07-114, IPEC Letter to USNRC, 10 CFR 50.59(d) Report for Indian Point Unit No. 3, dated 10/1/07

NL-09-135, IPEC Letter to USNRC, 10 CFR 50.59(d) Report for Indian Point Unit No. 3, dated 10/15/09

Drawings
501616, Primary Auxiliary Building Internal Flood Control Drain Piping, Rev. 0
9321-F-11663, PAB Concrete Floor Plan, Rev 16
9321-F-25283 DRN-07-01505, Containment Building Composite Piping at Reactor Coolant Pump No. 33, dated 3/16/07
9321-F-27363 Sh. 1, Flow Diagram Chemical & Volume Control System, Rev 51
9321-F-27503 Sh. 2, Flow Diagram Safety Injection System, Rev 50
9321-F-30083, Single Line Diagram D.C. System, Rev 58
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3-ARP-049, Panel Local – Intake Structure, Rev. 5
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3-PC-R73B, Nuclear Instrument Intermediate Range N-36 Channel Calibration, Rev. 7
3-PMP-012-SWS, Service Water Pump Removal and Installation, Rev. 19
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LIST OF ACRONYMS

AC	Alternating Current
ABFW	Auxiliary Boiler Feedwater
ADAMS	Agency-Wide Documents Access and Management System
AOP	Abnormal Operating Procedure
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CVCS	Chemical and Volume Control System
DBD	Design Basis Document
DC	Direct Current
D/P	Differential Pressure
DRN	Document Revision Notice
DRS	Division of Reactor Safety
EC	Engineering Change
ECN	Engineering Change Notice
EDG	Emergency Diesel Generator
EQ	Environmental Qualification
ERCN	Engineering Request Change Notice
FCV	Flow Control Valve
HELB	High Energy Line Break
ISI	In-Service Inspection
IST	In-Service Test
JW	Jacket Water
MDAFW	Motor Driven Auxiliary Feedwater
NEI	Nuclear Energy Institute
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission

PAB	Primary Auxiliary Building
PARS	Publicly Available Records
PMT	Post Maintenance Test
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank
SSC	Structure, System, and Component
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VC	Vapor Containment